2.1 Tier 1 and Tier 2* Departures from the DCD

The following Tier 1 and Tier 2* departures result from a change in the design described in the DCD.

STD ~~STP~~ DEP T1 2.1-1 SRV Setpoints and Simmer Margin

Description

The Reactor Safety/Relief Valve (SRV) relief and safety analytical limits and setpoints have been modified relative to Section 2.1.2 of the reference ABWR DCD (Tier 1 change) and also in DCD Sections 5.2 and 15.1 (Tier 2 changes).

These departures have been made for the following reasons:

<u>Tier 1 departures:</u>

- The setpoints and analytical limits have been increased to assure SRV simmer margin is not less than 15%. Simmer margin is the difference between the normal operating static pressure and the SRV direct actuation pressure based on nominal settings. The specific nameplate spring pressure and ASME rated flow rates at 103% of spring set pressure are increased for all valves.

<u>Associated Tier 2 departures:</u>

- The RCPB over-pressure protection analysis is revised to reflect the flow rates for the design SRVs rather than the SRVs described in the reference ABWR DCD.
- The reseating pressure (percent of spring setpoint) is revised to 96 - 90% to reflect the hardware capability. A requirement that the first two valve groups must reset at a value less that 92% is imposed as a result of the transient analysis performed to confirm that the containment hydrodynamic loads assumption of a single valve opening can occur multiple times after the initial multi-valve pop.
- The drift values are revised (increased) for both the safety and relief functions and are adjusted to 3% of the analytic limit to reflect operating experience. Note that all operating BWRs in the US that use standard Technical Specifications have adopted this 3% reset requirement for testing of SRVs.

The changes are required to meet the assumptions in the containment design and to reflect intended hardware design choices. They also maintain consistency with best practices in the US nuclear industry to increase system performance margin and to reduce maintenance requirements. This departure is expected to reduce unnecessary SRV testing during outages.

Evaluation Summary

This evaluation covered Tier 1 and Tier 2 departures.

- The changes to setpoints, reseating pressure, and drift values improve the reliability of the Safety Relief Valves.
- Analysis confirms that both reactor over pressurization and containment loading analyses are not adversely affected by the changes.

This departure was evaluated per Section VIII.A.4 of Appendix A to 10 CFR Part 52, which requires that 1) the design change will not result in a significant decrease in the level of safety otherwise provided by the design; 2) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; 3) special circumstances are present as specified in 10 CFR 50.12(a)(2); and 4) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. As shown below, each of these four criteria are satisfied.

(1) As discussed above, the design change represents an improvement and therefore will not result in a significant decrease in the level of safety otherwise provided by the design.

(2) The exemption is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. As discussed above, the design change represents an improvement and therefore will not present an undue risk to the public health and safety. The design change does not relate to security and does not otherwise pertain to the common defense and security.

(3) Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (iv) is present, since the design change represents an improvement based on BWR operating experience with SRVs. The analytically driven changes are the result of operating history and methodology refinement and therefore will result in a benefit to the public health and safety.

(4) The special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. Specifically, the design change represents an improvement in safety, and does not affect the configuration of the plant or the manner in which the plant is operated. Therefore, the reduction in standardization resulting from the change in the setpoints and other values should not adversely affect safety.

As demonstrated above, this exemption complies with the requirements in Section VIII.A.4 of Appendix A to 10 CFR Part 52. Therefore, STPNOC requests that the NRC approve this exemption.

STD DEP T1 2.2-1 Control Systems Changes to Inputs, Tests, and Hardware

Description

Minor changes are made to the safety-related and ~~non-safety-related~~ nonsafety-related control systems as described in the reference ABWR DCD. Design detailing, operational experience, technological improvement efforts, and the desire for clarity prompt the changes to the specific wording in Tier 1 ~~section~~ Section 2.2.

The reference ABWR DCD provides for an input to the Reactor Protection System (RPS) from Turbine First Stage Pressure. This protection sensor and signal will be replaced with a simulated thermal power signal of reactor power from the Neutron Monitoring System (NMS). The NMS will provide an accurate input of reactor power, regardless of the steam flow or steam bypass paths from the Main Steam System, that are not measured by first stage turbine pressure. This parameter provides a power threshold for a reactor trip on closure of a turbine trip valve or turbine control valve. This change will reduce installation costs, hardware failures, and operating costs over the life of the plant.

This departure provides for reactor power input to the RPS logic that generates a reactor trip signal on Turbine Stop Valve and Turbine Control Valve closure when reactor power is greater than 40%. The input from turbine first stage pressure will be eliminated, and replaced with a signal from the Simulated Thermal Power signal from the Neutron Monitoring System. The replacement of the input from turbine first pressure with a signal from neutron monitoring will be more accurate and more reliable and hence an improvement. With past and current experience of this logic, operating experience shows the benefit of this change to the plant design. This design has operational experience at Leibstadt (Switzerland - BWR 6), which was implemented approximately 20 years ago.

Without the four pressure sensors for turbine first stage pressure, the failure of these components is eliminated as a possible fault input to RPS for the reactor trip logic. Using the Neutron Monitoring System provides more accurate measurement of reactor power, by eliminating the steam flow paths that bypass the turbine (e.g., steam driven pumps, steam jet air ejectors, steam re-heaters, etc.). This method of reactor power measure input to RPS eliminates the mechanical measurement and possibility of mechanical failure of the turbine first stage instrumentation.

The original DCD Tier 1 Section 2.2.1 did not identify that the rod withdrawal block function associated with detection of the separation condition of a control rod is only applicable when the RPS Mode switch is in Startup Mode or Run Mode. Also, DCD Tier 1 Figure 2.2-1 did not identify that the RCIS receives more than the Refuel Mode status signal associated with the RPS Mode switch status (i.e. RCIS actually receives separate status signals associated with each of the four possible RPS Mode switch status conditions: 1) Shutdown Mode, 2) Refuel Mode, 3) Startup Mode and 4) Run Mode). When the RPS Mode switch is in Shutdown Mode, a rod withdrawal block is activated for all control rods; therefore, implementation of an individual rod withdrawal block based upon detection of the separation condition is not necessary.

When the RPS Mode switch is in Refuel Mode, it is only possible to withdraw: 1) one operable control rod or 2) the one or two control rods associated with a single HCU when the Scram Test mode of RCIS is active. All other operable control rods must remain fully inserted (and RCIS interlock logic enforces this situation). Thus, the RCIS logic insures the reactor remains in the subcritical condition regardless of the position of the one or two control rods that can be withdrawn with the RPS mode switch in Refuel Mode. When performing the FMCRD coupling check surveillance test in Refuel mode (for one or two control rods that have been withdrawn), the separation status will be activated when the FMCRD ball nut is withdrawn to the over travel out position. It is required that the separation rod withdrawal block not be activated to allow completion of this required surveillance test. Therefore, implementation of an individual rod withdrawal block based upon detection of the separation condition is also not desired and not necessary when in the Refuel Mode.

Design detailing efforts have located the ~~non-safety-related~~ nonsafety-related Feedwater Control System microprocessor-based equipment in both the Turbine Building and the Control Building. Therefore, the DCD Feedwater Control System Tier 1 description is changed to delete the sentence stating, “The FDWC System microprocessors are located in the Control Building.”

The reference ABWR DCD Tier 1 Table 2.2.1 ITAAC Acceptance Criteria for Item 11 (i.e. associated with testing of one of the dual redundant non-Class 1E uninterruptible power supply at a time) states the “test signal exists in only one channel at a time.” This acceptance criterion was based upon an assumption that in the RCIS design implementation each channel of the dual-redundant RCIS controller equipment would receive power from only one associated uninterruptible power supply. However, in the final RCIS design implementation, only the power supply associated with the one non-Class 1E uninterruptible power supply being tested will become inoperable and both of the dual-redundant controller channels remain operational when this testing is conducted. The detailed RCIS design for the dual-redundant controller equipment is implemented such that each channel remains operational as long as either one of the uninterruptible power supplies is operational. There is an associated alarm condition activated when one of the uninterruptible power supplies becomes inoperable (i.e. so the operator becomes aware of this abnormal power supply status condition). A change has been incorporated regarding the DCD Tier 1 ITAAC requirement for the RCIS related to the Acceptance Criteria associated with the testing of one of the dual redundant non-Class 1E uninterruptible power supply at a time.

Evaluation Summary

This departure was evaluated per Section VIII.A.4 of Appendix A to 10 CFR Part 52, which requires that 1) the design change will not result in a significant decrease in the level of safety otherwise provided by the design; 2) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; 3) special circumstances are present as specified in 10 CFR 50.12(a)(2); and 4) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. As shown below, each of these four criteria are satisfied.

(1) As discussed above, the design change represents an improvement and therefore will not result in a significant decrease in the level of safety otherwise provided by the design.

(2) The exemption is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. As discussed above, the design change represents an improvement and therefore will not present an undue risk to the public health and safety and the design change does not relate to security and does not otherwise pertain to the common defense and security.

(3) Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (iv) is present, since the design change represents an improvement and therefore will result in a benefit to the public health and safety.

(4) This is "standard" departure that is intended to be applicable to COL applicants that reference the ABWR DCD. Therefore this departure will not result in any loss of standardization, Additionally, the design change represents an improvement in safety, and does not adversely affect the configuration of the plant or the manner in which the plant is operated.

As demonstrated above, this exemption complies with the requirements in Section VIII.A.4 of Appendix A to 10 CFR Part 52. Therefore, STPNOC requests that the NRC approve this exemption.

STD DEP T1 2.3-1 Deletion of MSIV Closure and Scram on High Radiation

Description

The Scram and MSIV automatic closure on high MSLRM (main steam line radiation monitor) trip is deleted. Elimination of these functions reduces the potential for unnecessary reactor shutdown caused by spurious actuation of the MSLRM trip and increases plant operational flexibility. As a result, this change increases reliability and plant availability and therefore results in a benefit to public health and safety.

This departure includes the following Tier 1, Tier 2 and Technical Specification changes.

Tier 1 departures:

Changes have been made relative to the reference ABWR Tier 1 DCD Figure 2.3.1, “Process Radiation Monitoring System Control Interface Diagram” to remove the MSL Tunnel Area Radiation input from the plant sensors that provide input data.

Tier 2 departures:

Changes have been made relative to the reference ABWR Tier 2 DCD Sections 5.2, 6.2, 7.2, 7.3 and their associated tables to remove information pertaining to main steam line high radiation monitoring and process radiation monitoring system. Section 11.5 has been modified to move main steam line tunnel area radiation monitoring information from the section describing “monitoring required for safety and protection” to the section describing “monitoring required for plant operation.” Main steam line high radiation information has also been removed from Tables 18F-1 through ~~-3~~ 18F-3 and Tables 18H-2 and 18H-5.

Generic Technical Specification departures:

Generic Technical Specifications LCO 3.3.1.1 and LCO 3.3.6.1 and their associated Bases has been modified to remove the Main Steam Tunnel Radiation High function.

The original purpose of this instrumentation was to close the MSIVs in order to mitigate the potential release of fission products released from fuel rods. The initial release of primarily noble gas fission products from the damaged fuel rods was expected to cause a spike in the radiation readings on the steam line which would initiate safety related actions. However, radiation sources in the steam lines are primarily dominated by N-16 emissions, and setpoints sufficient to sense noble gas spikes can be overwhelmed by minor variations in N-16 flow causing spurious trips. Since sensors on the condenser steam jet air ejector and ventilation stack can also serve the purpose of monitoring potential offsite releases, the BWROG LTR NEDO-31400A concluded that the vessel isolation (MSIVs) and scram functions of the MSLRM are not required.

The MSLRM alarms in the main control room and the conclusion of NEDO-31400A ~~section~~ Section 2.2 remain valid for the STP proposed design. Both the MSLRM and the Condenser Steam Jet Air Ejector (SJAE) monitors will alert the operators for events that can cause an elevated release of radioactivity from the fuel. In addition to these alarms, given sufficient time, the offgas treatment system radiation monitor and eventually the stack effluent monitor will activate alarms in the control room and the operator will be able to isolate the offgas system to stop these releases to the environment. Therefore, even without the automatic reactor shutdown function and the MSIV closure on the MSLRM trip, the operator will be able to limit offsite releases.

For operating plants, the NRC has approved this change (see NEDO-31400A). The following additional considerations apply to the ABWR:

In the NEDO Section 9 there was a discussion of the negligible increase in reactivity control failure frequency with the deletion of the MSLRM scram function. This increase was due to operating plants taking credit for the MSLRM initiating a scram during the control rod drop accident. As described in Tier 2 DCD Section 15.4.10, it is concluded that for the ABWR, there is no basis for the control rod drop event to occur or to perform a related radiological analysis. The deletion of the automatic scram and MSL isolation result in no change in associated risk.

The NRC's approval letter accepting the NEDO requires an applicant referencing it to:

- Demonstrate that the plant design features affecting a rod drop analysis bound those used in the report. For ABWR, since the event has no basis. A comparison of design features affecting the rod drop analysis against those assumed in the report is unnecessary.
- Demonstrate a basis (e.g., plant procedures) to conclude that manual operator action will be taken expeditiously in case of increased levels of radioactivity in the MSLs. STP 3 & 4 alarm response procedures will direct operator action in case of increased radioactivity levels.

Since this change incorporates a BWR change that was previously approved by the NRC and since the SER conditions are met for the ABWR as explained above, there are no adverse effects on plant performance.

Evaluation Summary

This departure was evaluated per Section VIII.A.4 of Appendix A to 10 CFR Part 52, which requires that 1) the design change will not result in a significant decrease in the level of safety otherwise provided by the design; 2) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; 3) special circumstances are present as specified in 10 CFR 50.12(a)(2); and 4) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. As shown below, each of these four criteria are satisfied.

(1) As discussed above, the design change represents an improvement and therefore will not result in a significant decrease in the level of safety otherwise provided by the design.

(2) The exemption is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. As discussed above, the design change represents an improvement and therefore will not present an undue risk to the public health and safety. The design change does not relate to security and does not otherwise pertain to the common defense and security.

(3) Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (iv) is present, since the design change represents an improvement and therefore will result in a benefit to the public health and safety.

(4) This is “standard” departure that is intended to be applicable to COL applicants that reference the ABWR DCD. Therefore this departure will not result in any loss of standardization, Additionally, the design change represents an improvement in safety, and does not adversely affect the configuration of the plant or the manner in which the plant is operated.

As demonstrated above, this exemption complies with the requirements in Section VIII.A.4 of Appendix A to 10 CFR Part 52. Therefore, STPNOC requests that the NRC approve this exemption.

References

(1) Licensing Topical Report NEDO-31400A “Safety Evaluation for Eliminating the BWR Main Steam Isolation Valve Closure Function and the Scram Function of the Main Steam Line Radiation Monitor,” October 1992.

STD DEP T1 2.4-1 Residual Heat Removal System and Spent Fuel Pool Cooling

Description

The reference ABWR DCD has two RHR loops connected to the Fuel Pool Cooling system with normally closed crosstie valves. During refueling outages, a crosstie valve can be opened to allow direct cooling of the fuel pool by circulation of fuel pool water through the RHR heat exchanger and returning it to the fuel pool. In addition, the RHR pumps have the capability to provide fuel pool emergency makeup water by transferring suppression pool water to the fuel pool. This change is to add the capability to allow the choice of a third loop, RHR division A, in the Augmented Fuel Pool Cooling and Fuel Pool Makeup Modes.

This addition of piping and valves will be of the same quality standard, seismic category, and ASME code as the B and C RHR loops components, along with another capability to provide makeup or cooling to the Spent Fuel Pool. Only one RHR cooling loop will be aligned for the Augmented Fuel Pool Cooling or Fuel Pool Makeup Mode at any one time. The additional loop will increase the reliability from a single failure standpoint. This design change was chosen based on improved reliability and performance.

This change provides the ability to supply fuel pool cooling or makeup from any of the three RHR loops in the Augmented Fuel Pool Cooling or Fuel Pool Makeup Modes. This will enhance capabilities and reliability to perform division outages for maintenance and other activities. Division outages will be better able to be coordinated during all plant operational Modes. During design detailing it was recognized that the added flexibility of having the capability to perform divisional outages in any order was a worthwhile design improvement. As an example, if Division B EDG constitutes a critical path for an outage, in order to maintain a single failure margin, work could not start until core decay heat decreased to the point that RHR Spent Fuel Pooling augmented cooling was no longer required. By having all three divisions capable of supporting the Augmented Fuel Pool Cooling Mode, Divisional Outages (potential critical path) could occur based on workload in the division.

Evaluation Summary

During design detailing it was recognized that the added flexibility of having the capability to perform divisional outages in any order was a worthwhile design improvement. As an example, if Division B EDG constitutes a critical path for an outage, in order to maintain a single failure margin, work could not start until core decay heat decreased to the point that RHR Spent Fuel Pooling Assist was no longer required. By having all three divisions capable of supporting Spent Fuel Pool Cooling assist, Divisional Outages (potential critical path) could occur based on workload in the division.

This departure was evaluated per Section VIII.A.4 of Appendix A to 10 CFR Part 52, which requires that 1) the design change will not result in a significant decrease in the level of safety otherwise provided by the design; 2) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; 3) special circumstances are present as specified in 10 CFR 50.12(a)(2); and 4) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. As shown below, each of these four criteria are satisfied.

(1) As discussed above, the design change represents an improvement and therefore will not result in a significant decrease in the level of safety otherwise provided by the design.

(2) The exemption is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. As discussed above, the design change represents an improvement and therefore will not present an undue risk to the public health and safety and the design change does not relate to security and does not otherwise pertain to the common defense and security.

(3) Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (iv) is present, since the design change represents an increase in redundancy and therefore will result in a benefit to the public health and safety.

(4) This is “standard” departure that is intended to be applicable to COL applicants that reference the ABWR DCD. Therefore this departure will not result in any loss of standardization, Additionally, the design change represents an improvement in safety, and does not adversely affect the configuration of the plant or the manner in which the plant is operated.

As demonstrated above, this exemption complies with the requirements in Section VIII.A.4 of Appendix A to 10 CFR Part 52. Therefore, STPNOC requests that the NRC approve this exemption.

STD DEP T1 2.4-2 Feedwater Line Break Mitigation

Description

This departure reduces challenges to the containment pressure design value following a feedwater line break (FWLB). The corrective design concept is a trip of the condensate pumps following an indication that a Feedwater Line Break (FWLB) in the drywell has occurred.

The FWLB is the limiting design basis accident for ABWR primary containment vessel (PCV) peak pressure response. This is because blowdown flows from both the reactor pressure vessel (RPV) side and the balance of plant (BOP) feedwater side contribute to the peak pressure response. Previous BWR designs were bounded by the recirculation line break, that is not a consideration in the ABWR design.

Calculations performed during STP 3 & 4 initial planning as a check against the FSER estimate that the ABWR containment design pressure would be exceeded at the 3926 MWt power level with an injection of approximately 100% feedwater flow at 15 minutes. The licensing basis for ABWR is no operator actions for 30 minutes for design basis accidents. With the current ABWR design, the only mitigation would be operator action using the non-safety trip of the condensate and/or feedwater pumps.

Therefore, high drywell pressure signals already existing in the Leak Detection & Isolation (LDI) logic of the Safety System Logic & Control (SSLC) are used, in conjunction with differential pressure signals between the two feedwater lines, to identify a FWLB in containment and to then trip the condensate pumps.

The departure implementation of condensate pump trip improves plant safety by limiting the mass flow to the drywell after the FWLB, thereby ensuring the predicted peak pressure will not exceed the design value. The instrumentation logic to initiate the trip will be an “AND” circuit to reduce the probability of false trips. That is, the logic will require excessive differential pressure between the two-feedwater lines “AND” high drywell pressure to initiate the condensate pump trip. This will reduce the negative impact on plant operation, plant reliability and availability. There would not be an impact on the PRA by adding circuit breakers for the condensate pump supplies because the logic will only be initiated during FWLB LOCA, the breakers will be normally closed, and additional operator actions will not be required to start the condensate pumps during other events.

Evaluation Summary

These changes ensure that the containment pressure margins are maintained during the limiting containment pressurization accident. Consequentially, the changes decrease the risk associated with the feedwater line break inside containment. These changes maintain the same level of plant reliability and performance as described in the DCD. The changes will provide a better level of plant protection and personal safety and a net benefit to the public health and safety. While this involves changes to an SSC, there are no adverse effects on any DCD design function. No procedure was changed.

This departure was evaluated per Section VIII.A.4 of Appendix A to 10 CFR Part 52, which requires that 1) the design change will not result in a significant decrease in the level of safety otherwise provided by the design; 2) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; 3) special circumstances are present as specified in 10 CFR 50.12(a)(2); and 4) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. As shown below, each of these four criteria are satisfied.

(1) As discussed above, the design change represents an improvement and therefore will not result in a significant decrease in the level of safety otherwise provided by the design.

(2) The exemption is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. As discussed above, the design change represents an improvement and therefore will not present an undue risk to the public health and safety. The design change does not relate to security and does not otherwise pertain to the common defense and security.

(3) Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (iv) is present, since the design change represents an improvement and better conformance to licensing criteria (no operator action until 30 minutes) and therefore will result in a benefit to the public health and safety.

(4) This is “standard” departure that is intended to be applicable to COL applicants that reference the ABWR DCD. Therefore this departure will not result in any loss of standardization. Additionally, the design change represents an improvement in safety, and does not adversely affect the configuration of the plant or the manner in which the plant is operated.

As demonstrated above, this exemption complies with the requirements in Section VIII.A.4 of Appendix A to 10 CFR Part 52. Therefore, STPNOC requests that the NRC approve this exemption.

STD DEP T1 2.4-3 RCIC Turbine/Pump

Description

The original DCD incorporated a steam turbine driven water pump that has been historically used in the United States with BWR plants. During the design detailing stage of the ABWR development, another design was chosen based on improved reliability, performance, and simplicity. The new design meets or exceeds all safety-related system performance criteria including start time, flow rate, and low steam pressure operation.

The improved design and system simplification is due to (a) monoblock design (pump ~~&~~ and turbine within same casing); (b) no shaft seal required; (c) no barometric condenser required; (d) no oil lubrication or oil cooling system required because the system is totally water lubricated; (e) no steam bypass line required for startup; (f) simpler auxiliary subsystems; and (g) no vacuum pump and associated penetration piping or isolation valves required. The monoblock design is of horizontal, two-stage centrifugal water pump driven by a steam turbine contained in a turbine casing integral with the pump casing. The turbine wheel has a single row of blades. The pump impellers, turbine wheel and inducer are mounted on a common shaft, which is supported on two water lubricated journal bearings. The bearings are housed in a central water chamber between the turbine and pump sections and are lubricated by a supply of water taken from the discharge of the first stage impeller and led to the bearings through a water strainer.

The pump is supported on the pedestals of a fabricated steel base plate by feet formed on the pump casing and central water chamber. The monoblock construction of the pump eliminates the need for alignment between the pump and the turbine. The operating speed of the pump is governed by the turbine control subsystem which regulates the quantity of steam to the turbine based on discharge pressure. The main elements of the control gear are the steam stop valve, the throttle valve and the pressure governor. The pump is also provided with electrical and mechanical over speed trip mechanisms which close the steam stop valve when the speed exceeds predetermined levels. Speed measurement is provided by an electronic tachometer.

One less containment penetration is required and approximately 10 meters of small bore piping previously analyzed for interfacing system LOCAs and upgraded to burst pressure have been removed from the design. The fire loading in the RCIC pump room is reduced by the elimination of the lube oil subsystem and 106 liters of Class III B lube oil.

Licensing Topical Report NEDE-33299P was submitted to the NRC by General Electric Company December 2006 proposing this change as a generic revision to the Design Control Document. More detail on this change may be found in this LTR.

Evaluation Summary

The events in FSAR Chapter 15 were evaluated based on the quickened response times expected so the dynamics of upset and accident responses are not compromised. The events and accidents in Chapter 15 were reviewed. The analyses and conclusions presented in Chapter 15 are not affected. No negative impacts on severe accident probability or severity have been identified nor has a new type of severe accident been created. The bases in the generic Technical Specifications in Chapter 16 will be met or exceeded. This departure results in no negative impact on safety, plant operation or cost. Plant availability and reliability will improve due reduction of active and passive components. Improved turbine reliability well improves plant safety as will improve transient and startup characteristics.

This departure was evaluated per Section VIII.A.4 of Appendix A to 10 CFR Part 52, which requires that 1) the design change will not result in a significant decrease in the level of safety otherwise provided by the design; 2) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; 3) special circumstances are present as specified in 10 CFR 50.12(a)(2); and 4) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. As shown below, each of these four criteria are satisfied.

(1) As discussed above, the design change represents an improvement and therefore will not result in a significant decrease in the level of safety otherwise provided by the design.

(2) The exemption is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. As discussed above, the design change represents an improvement and therefore will not present an undue risk to the public health and safety. The design change does not relate to security and does not otherwise pertain to the common defense and security.

(3) Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (iv) is present, since the design change represents an improvement and therefore will result in a benefit to the public health and safety.

(4) This is “standard” departure that is intended to be applicable to COL applicants that reference the ABWR DCD. Therefore this departure will not result in any loss of standardization. Additionally, the design change represents an improvement in safety, and does not adversely affect the configuration of the plant or the manner in which the plant is operated.

As demonstrated above, this exemption complies with the requirements in Section VIII.A.4 of Appendix A to 10 CFR Part 52. Therefore, STPNOC requests that the NRC approve this exemption.

References

(1) NEDE-33299P, “Licensing Topical Report - Advanced Boiling Water Reactor (ABWR) with Alternate RCIC Turbine-Pump Design,” December, 2006

STD DEP T1 2.12-1 Electrical Breaker/Fuse Coordination and Low Voltage Testing

Description

The reference ABWR DCD in Tier 1 states electrical power distribution interrupting devices (circuit breakers and fuses) are coordinated such that the interrupting device closest to the fault opens first. The description of the interruption device coordination has been modified to include the acceptable industry practice with standards and codes (e.g., IEEE 141, IEEE 242, etc.). Including this provides detailed guidance for electrical system design expectations. Since protective device coordination may overlap, and the discrete coordination may not be possible, the expectation has been changed to meet the requirement to the maximum extent possible.

The reference ABWR DCD ITAAC also requires that pre-operational/start-up testing of the as-built Class 1E Electrical Power Distribution System will be conducted by operating connected Class 1E loads at their analyzed minimum voltage. DCD Table 2.12.1 (Electric Power Distribution System ITAAC) currently states that tests of the as-built Class 1E Electric Power Distribution System will be conducted by operating connected Class 1E loads at their analyzed minimum voltage. Testing in this manner for each connected Class 1E load is not practical to connect and disconnect each load, one at time to facilitate testing.

For DC loads, ITAAC require testing by operating connected Class 1E loads at both the minimum and maximum battery voltages. Tier 1 DCD Table 2.12.12 (Direct Current Power Supply ITAAC) currently states that tests of the as-built Class 1E DC system will be conducted by operating connected Class 1E loads at less than or equal to the minimum allowable battery voltage and at greater than or equal to the maximum battery charging voltage. It is not practical to perform testing in this manner. This is modified to allow performance type tests at the manufacturer's shop for the operating voltage range of Class 1E AC and DC electrical equipment prior to shipment to the site. In addition, system preoperational tests will be conducted on the as-built Class 1E AC and DC systems and test voltage results will be compare against system voltage analysis.

Evaluation Summary

For electrical loads powered at or below 120 VAC or 125 VDC, the requirement that the device closest to the fault open first is not always met, since many small loads have internal fuses/circuit breakers and there is often a minimum device size available, or the minimum circuit breaker/fuse size recommended by the vendor. In the case of high fault current, the upstream protective device may trip before the protective device connected to the small load, or both may trip at the same time. In such cases, discrete coordination may not be possible.

The extensive in-situ testing in the DCD is not necessary and is duplicated, since the voltage tests are performed by the manufacturers as part of their normal performance and functional tests prior to shipment. In addition, testing is performed at the jobsite on electrical power distribution equipment during construction after its installation.

The events and accidents in Chapter 15 were reviewed. The analyses and conclusions presented in Chapter 15 are not affected as the alternate methods of breaker coordination and low voltage testing are judged equivalent to those in the DCD. No negative impacts on severe accident probability or severity have been identified nor has a new type of severe accident been created. The bases in the generic Technical Specifications in Chapter 16 will be met or exceeded.

This departure was evaluated per Section VIII.A.4 of Appendix A to 10 CFR Part 52, which requires 1) the design change will not result in a significant decrease in the level of safety otherwise provided by the design; 2) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; 3) special circumstances are present as specified in 10 CFR 50.12(a)(2); and 4) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. As shown below, each of these four criteria are satisfied.

(1) As discussed above, the change is intended to accomplish the same purpose as the original DCD design and therefore will not result in a significant decrease in the level of safety otherwise provided by the design.

(2) The exemption is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. As discussed above, the DCD change accomplishes the same purpose and therefore will not present an undue risk to the public health and safety. and the design change does not relate to security and does not otherwise pertain to the common defense and security.

(3) Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (ii) is present, since the change accomplishes the same underlying purpose as the original DCD design.

(4) This change is intended to be applicable to COL applicants that reference the ABWR DCD. Therefore this departure will not result in any loss of standardization.

As demonstrated above, this exemption complies with the requirements in Section VIII.A.4 of Appendix A to 10 CFR Part 52. Therefore, STPNOC requests that the NRC approve this exemption.

References

(1) IEEE 141-1993, Recommended Practice for Electric Power Distribution for Industrial Plants (IEEE Red Book)

(2) IEEE 242 -2001, Recommended Practice for Protection and Coordination of Industrial and Commercial Power Systems (IEEE Buff Book)

STD DEP T1 2.12-2 I&C Power Divisions

Description

A fourth division of safety related power has been added to the Class 1E Instrument and Control Power Supply System.

The Instrument and Control Power Supply System as described in the DCD Tier 1 provided power to three mechanical safety-related divisions (I, II and III) and not to safety-related Distributed Control and Information System (DCIS) Division IV. This departure adds a fourth regulating transformer and associated distribution panels to supply Instrument and Control Power to Division IV.

The DCIS cabinets and chassis, ECCS Digital Control and Information System cabinets and chassis, in each of the four divisions, use redundant power supplies and feeds for increased reliability and availability to allow self-diagnostics and to operate during power failures. The existing design provides three divisions such that the two feeds are uninterruptible vital AC power (uninterruptible does not mean single failure proof) and I&C power (interruptible but diesel-backed). The second I&C power feed is available to the Division IV DCIS cabinets and chassis. Most power problems can be addressed on-line and all such problems will be “non-critical” faults since no functionality will be lost.

Evaluation Summary

This departure was evaluated per Section VIII.A.4 of Appendix A to 10 CFR Part 52, which requires 1) the design change will not result in a significant decrease in the level of safety otherwise provided by the design; 2) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; 3) special circumstances are present as specified in 10 CFR 50.12(a)(2); and 4) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. As shown below, each of these four criteria are satisfied.

(1) As discussed above, the design change represents an improvement and therefore will not result in a significant decrease in the level of safety otherwise provided by the design.

(2) The exemption is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. As discussed above, the design change represents an improvement and therefore will not present an undue risk to the public health and safety and the design change does not relate to security and does not otherwise pertain to the common defense and security.

(3) Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (iv) is present, since the design change represents an improvement and therefore will result in a benefit to the public health and safety.

(4) This is “standard” departure that is intended to be applicable to COL applicants that reference the ABWR DCD. Therefore this departure will not result in any loss of standardization. Additionally, the design change represents an improvement in safety, and does not adversely affect the configuration of the plant or the manner in which the plant is operated.

As demonstrated above, this exemption complies with the requirements in Section VIII.A.4 of Appendix A to 10 CFR Part 52. Therefore, STPNOC requests that the NRC approve this exemption.

STD DEP T1 2.14-1 Hydrogen Recombiner Requirements Elimination

Description

The departure relative to the reference ABWR Tier 1 and Tier 2 DCD and Generic Technical Specifications is documented in detail in Licensing Topical Report (LTR) NEDE-33330P, “Hydrogen Recombiner Requirements Elimination”, March 2007, proposing this change as a generic revision to the Design Control Document. More detail on this change may be found in the LTR.

The reference ABWR DCD requires two redundant hydrogen recombiners and safety-related hydrogen/oxygen analyzers. This includes associated containment isolation valves, safety-related cooling water, and Class 1E power supply. This departure removes the hydrogen recombiners and associated components. The hydrogen and oxygen monitors are retained but downgraded from safety-related to ~~non-safety-related~~ nonsafety-related. This change will not affect the Containment Spray System and the mixing it provides to prevent oxygen pockets.

Amended since DCD issuance, 10 CFR 50.44, “Standards for Combustible Gas Control System in Light Water-Cooled Power Reactors”, does not currently require light water reactors, operating with an inerted containment, to have hydrogen recombiners. With this rule change, the recombiners and hydrogen monitoring equipment no longer meets any of the criteria in 10 CFR 50.36(c)(2)(ii) for retention in the Technical Specifications and are removed from Chapter 16 and Part 4.

Evaluation Summary

This departure was evaluated per Section VIII.A.4 of Appendix A to 10 CFR Part 52, which requires that 1) the design change will not result in a significant decrease in the level of safety otherwise provided by the design; 2) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; 3) special circumstances are present as specified in 10 CFR 50.12(a)(2); and 4) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. As shown below, each of these four criteria are satisfied.

(1) As discussed above, the design change conforms to current regulations and therefore will not result in a significant decrease in the level of safety otherwise provided by the design.

(2) The exemption is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. As discussed above, the design change does not present an undue risk to the public health and safety. The design change does not relate to security and does not otherwise pertain to the common defense and security.

(3) Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (ii), and special circumstance (vi) are invoked as evidenced by the revision to 10 CFR 50.44 as the underlying purpose is still served and the revision of regulations is a material change of circumstances.

(4) This is "standard" departure that is intended to be applicable to COL applicants that reference the ABWR DCD. Therefore this departure will not result in any loss of standardization.

As demonstrated above, this exemption complies with the requirements in Section VIII.A.4 of Appendix A to 10 CFR Part 52. Therefore, STPNOC requests that the NRC approve this exemption.

References

(1) Licensing Topical Report NEDE-33330P, "Hydrogen Recombiner Requirements Elimination" March 2007

(2) 10 CFR 50.44, "Standards for Combustible ~~gas~~ Gas Control System in Light Water-Cooled Power Reactors"

STD DEP T1 3.4-1 Safety-Related I&C Architecture

Description

~~This departure modifies the design of certain devices, functions and standards related to the Essential Multiplexing System (EMS) and Safety System Logic and Controls (SSLC). In general, this collection of changes enable the descriptions of the EMS and SSLC to be modified in such a way that they describe integrated top-level functions with a set of supporting sub-functions, as opposed to the current DCD descriptions, which describe them as separate systems with single-purpose hardware components.~~

~~The reference ABWR DCD design descriptions reflect outdated technology and are inconsistent with currently available systems and equipment. This change deletes references to components such as Control Multiplexing Units (CMUs), Remote Multiplexing Units (RMUs) and others that refer to an outdated technology (multiplexing), and imply hardware components (units) with limited purposes. In the text changes in FSAR chapter 7 and elsewhere, equivalent data communication functions are described, but not in the context of "multiplexing" or specific hardware components.~~

~~This departure also enables specific architecture changes in the Engineered Safety Functions (ESF) portion of the I&C architecture. Specifically, it limits the application of dual redundant Safety System Logic Units (SLUs) and 2-out-of-2 output voting only to those situations where the physical system arrangement and consequences of inadvertent actuation of equipment warrant such protection against inadvertent actuations. Also it eliminates SLU channel bypass function in this same portion of the structure. Also it allows the use of multiple processors where figures imply single processors.~~

~~This departure also deletes or supplements references to specific outdated communication protocol standards (both Tier 2* and Tier 2.) In the ten plus years that have passed since the reference ABWR DCD was finalized, network technologies have evolved to the point where the concepts and hardware described in the DCD are no longer available in modern, commercially available networks.~~

~~At the time the GE ABWR was certified, the detailed design of the I&C equipment was not established. GE recognized that with the rapid evolution of the I&C technology, the preferred I&C design, including the design of equipment implementing the logic of ESF systems, would almost certainly include modules, components and capability not yet envisioned. Consequently, a specific equipment design for the SSLC systems was not established.~~

~~Even though no specific I&C equipment design was established for the ABWR, GE understood that the ABWR would make more extensive use of digital equipment than any previous plant design. GE developed a design of the SSLC and the supporting Essential Multiplexing System (EMS) using then current technology. These were structured as separate systems.~~

~~This generalized design, particularly for the EMS, was necessary to establish fundamental architectural elements, provides for the adequacy of diversity, and establishes a sufficiently comprehensive set of standards to be applied for the actual detail design. Specifically, such aspects as the system architecture constraints, system and software design processes, and equipment qualification requirements were established. These specific requirements relative to the I&C divisional architecture, inclusion of hardwired backups for diversity, specific design process related standards to be followed, and specific EMS redundancy requirements became the ABWR DCD. All of those basic requirements are unaffected by the changes covered by this report, but in some cases the requirements are met with a different SSLC architecture.~~

~~GE established the architecture for the ABWR for both the RPS and ESF portions of the SSLC based on NUMAC-type equipment. NUMAC-based I&C typically uses one NUMAC chassis per division for the system unless the total input/output count is too large, or the total computational load exceeds the capacity of the NUMAC chassis processing modules, or there is a specific need to maintain operability of part of the channel with equipment out of service. At that time, multiplexed processing of data was typically handled with somewhat independent "multiplexing systems". Based on implementation with NUMAC-type equipment, the potential loading for specific processors, and a separate multiplexing system, the design divided the SSLC into such sub-parts as digital trip modules (DTMs), trip logic units (TLUs), safety system logic units (SLUs) and remote multiplexing units (RMUs) and included a separate EMS. The design for ESF included pairs of SLUs in each division. In addition to the SSLC design, a design of the EMS was established based on the then-available methods and standards in the rapidly advancing area of multiplexing. The ABWR certified design was used to evaluate overall system issues and was the basis for PRA evaluations.~~

~~The DCD requirements included: 1) I&C divisions and system logic assignment; 2) divisions of sensors (typically all four divisions); 3) divisions of actuators (three divisions for ESF systems); and 4) EMS architecture having redundancy within each division. The ABWR supporting Tier 2 material also included the full safety design bases for all of the SSLC-related systems and a description of how the General Design Criteria (GDC) are satisfied. None of these requirements is affected by the changes covered by this departure.~~

This departure can be characterized into three primary changes.

(1) Elimination of obsolete data communication technology

The departure eliminates references to the Essential Multiplexer System (EMS) and the Non-Essential Multiplexer System (NEMS) originally envisioned in the ABWR architecture and replaces them with separate and independent system level data communication capabilities. The original concept was based on a common EMS, which could be used by multiple safety-related, digitally-based protection systems. This departure defines separate dedicated data communication for each safety-related digital platform, including separate and independent data communication for each division within a system. The original concerns expressed by the NRC related to the common EMS are addressed as part of Appendix 7A and have been updated to reflect the separate communication capabilities.

In addition, the reference ABWR DCD identified use of the data communication standard ANSI-X3 series, Fiber Distributed Data Interface (FDDI), as the communication protocol for the EMS. FDDI is an obsolete technology and no longer appropriate for use. The safety-related data communication will use a combination of proprietary network data communication and dedicated point-to-point communication to fully meet the defined data communication functional requirements.

The elimination of the multiplexer concept required all references to the system(s) and its primary components to be replaced with a generic data communication reference. The terms EMS and NEMS were eliminated along with Remote Multiplexer Unit (RMU) and Control Room Multiplexer Unit (CMU).

The communication functions are described in FSAR Sections 7.2, 7.3 and 7.9S.

(2) Elimination of unnecessary inadvertent actuation prevention logic and equipment

The reference ABWR DCD described the design of the Engineered Safety Features (ESF) actuation outputs as being fully redundant within each division of the ESF digital controls systems. This design was to minimize the potential for false actuation of ESF components. In the design, each output was processed through two redundant sets of hardware and a final two-out-of-two (2/2) logic decision was to be performed on a component level. Both sets of outputs had to demand actuation before a component would actually respond. As part of the detailed design of the ABWR ESF digital controls, it was determined that only selected ESF components required the redundant actuation prevention logic. If actuated during normal plant operation, most of the ESF components do not have an adverse impact on the safety or operation of the plant. The limited set of components that cannot be actuated during normal operation, such as the main steam isolation valves, are provided with redundant actuation equipment and logic.

The complexity of implementing the fully redundant actuation logic was found to be a detriment to the design, and significantly increased the required maintenance and testing while providing no increase in true plant reliability. As a result, the redundant actuation logic is only implemented for components that may impact plant safety or operation if actuated during normal plant operation.

(3) Clarifications of digital controls nomenclature and systems

The reference ABWR DCD defined many functional design requirements in terms typically reserved for hardware. Examples include the terms “module,” “unit,” and “system.” the terminology was corrected to refer to the requirement as a “function.” The terminology was corrected to refer to the requirement as a “function” to eliminate the confusion associated with purely functional requirements and not physical requirements defined in the DCD. Examples include:

- Digital Trip Module (DTM) to Digital Trip Function (DTF)
- Trip Logic Unit (TLU) to Trip Logic Function (TLF)
- Safety System Logic Unit (SLU) to Safety System Logic Function (SLF)
- Plant Computer System (PCS) to Plant Computer Function (PCF)
- Essential Multiplexer System (EMS) to Essential Communication Function (ECF)

In addition, to better define the functional design and implementation of the digital controls platforms, specific I&C system names were assigned to the ESF digital controls systems and the Reactor Protection System (RPS). The digital controls responsible for the ESF systems are designated as the ESF Logic & Control System (ELCS). The RPS functions are implemented in two separate I&C systems: the Reactor Trip & Isolation System (RTIS) and the Neutron Monitoring System (NMS). The term Safety System Logic & Control (SSLC) was clarified as a general term used to cover all of the logic and controls associated with safety-related control systems.

The nomenclature changes required several sections of the original DCD to be updated for the STP 3&4 COLA to make all sections consistent.

Evaluation Summary

This departure was evaluated per Section VIII.A.4 of Appendix A to 10 CFR Part 52, which requires that 1) the design change will not result in a significant decrease in the level of safety otherwise provided by the design; 2) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; 3) special circumstances are present as specified in 10 CFR 50.12(a)(2); and 4) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. As shown below, each of these four criteria are satisfied.

(4) As discussed above, the design change represents another method for accomplishing the same purpose and therefore will not result in a significant decrease in the level of safety otherwise provided by the design.

(5) The exemption is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. As discussed above, the design change represents an improvement and therefore will not present an undue risk to the public health and safety. The design change does not relate to security and does not otherwise pertain to the common defense and security.

(6) Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, special circumstance (ii) is present, since the design change represents another method of accomplishing the underlying purpose of the DCD.

(7) This is “standard” departure that is intended to be applicable to COL applicants that reference the ABWR DCD. Therefore this departure will not result in any loss of standardization.

As demonstrated above, this exemption complies with the requirements in Section VIII.A.4 of Appendix A to 10 CFR Part 52. Therefore, STPNOC requests that the NRC approve this exemption.

STD DEP T1 5.0-1 Site Parameters

Description

The site parameters in the reference ABWR DCD were selected to bound most potential US sites. However, the STP 3 & 4 site, when site historical data is analyzed to using current methodologies and standards, represents three specific increases from the generic envelope.

The site design basis flood level is increased from that specified in the DCD. The certified design site parameter for site flooding is changed from 30.5 cm below grade to 414.5 cm above grade (1036.3 cm above mean sea level (MSL)) in order to handle a main cooling reservoir failure as a design basis event at STP.

The main cooling reservoir at the South Texas site is a non-seismic category 1 dam; hence, its failure must be assumed in the worst possible location. This results in the site design basis flood. The maximum flood level is 1450.8 cm above MSL; however it decreases with distance from the main cooling reservoir.

STP 3 & 4 safety-related SSCs are designed for or protected from this flooding event by watertight doors to prevent the entry of water into the Reactor Buildings and Control Buildings in case of a flood. Exterior doors located below the maximum flood elevation on the 12300 floor of the Reactor Building and Control Building are revised to be water-tight doors. The Ultimate Heat Sink storage basin and the RSW pump houses are water-tight below the flood level.

The maximum design precipitation rate for rainfall at the STP site is calculated to increase from 49.3 cm/hr to 50.3 cm/hr based on site meteorology studies. This value is one factor in determining the structural loading conditions for roof design. ABWR Seismic Category 1 structures have roofs without parapets or parapets with scuppers to supplement roof drains so that large inventories of precipitation cannot accumulate. Therefore, the increase in maximum rainfall rate does not result in a substantial increase in the roof design loading, and therefore does not affect the design of these structures.

The humidity at the STP 3 & 4 site, as represented by wet bulb temperature, is increased from that specified in the DCD.

Wet Bulb 1% Exceedance Values		DCD	STP 3 & 4
	Maximum Coincident	25°C	26.3°C
	Maximum Non-coincident	26.7°C	27.3°C
Wet Bulb 0% Exceedance Values (historical limit)			
	Maximum Non-coincident	27.2°C	29.1°C

The maximum dry-bulb temperature in combination with coincident wet-bulb temperature provides the state point (enthalpy of the air) that is used as design input for HVAC system design to determine cooling loads. The 1% exceedance STP site-specific state point value is not bounded by the 1% exceedance ABWR state point value.

The Control Building HVAC, Reactor Building Secondary Containment HVAC, and Reactor Building Safety Related Electrical Equipment HVAC systems are designed for an outdoor summer maximum temperature of 46°C. This temperature corresponds to the ABWR 0% exceedance value. The ABWR 0% exceedance state point bounds the STP site-specific 0% exceedance state point and the 1% exceedance state point. The reference ABWR DCD cooling loads calculated based on 0% exceedance values for Control Building HVAC, Reactor Building Secondary Containment HVAC, and Reactor Building Safety Related Electrical Equipment HVAC systems are bounding. Therefore, the change in 1% exceedance coincident wet bulb temperature has no adverse impact on these HVAC systems.

The Radwaste Building HVAC systems have been redesigned using STP site-specific ambient temperatures and the revised HVAC design is compliant with STP 3 & 4 Characteristics.

The maximum non-coincident wet-bulb temperature is used as input for short-term performance of cooling towers and evaporative coolers. In the case of STP 3 & 4, this value is an hourly data point. The site-specific maximum non-coincident wet-bulb temperatures on an hourly basis are not bounded by the reference ABWR site parameters. However, the calculated 30-day and 24-hour consecutive maximum non-coincident wet-bulb temperatures have been determined to be less than the reference ABWR DCD non-coincident hourly value. The UHS cooling tower long-term cumulative evaporation for the postulated LOCA case has been evaluated using the STP site-specific worst-case 30 consecutive day weather data. The UHS basin water temperature has been evaluated using the worst one-day (24 hour) weather data. Thus, the 0% exceedance and 1% exceedance values for non coincident wet-bulb temperatures not being bounded have no adverse impact on the STP 3 & 4 UHS analysis.

Evaluation Summary

These changes establish an equivalent level of site reliability and performance as described in the DCD.

This departure was evaluated per Section VIII.A.4 of Appendix A to 10 CFR Part 52, which requires that 1) the design change will not result in a significant decrease in the level of safety otherwise provided by the design; 2) the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security; 3) special circumstances are present as specified in 10 CFR 50.12(a)(2); and 4) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. As shown below, each of these four criteria are satisfied.

(1) As discussed above, the design change will maintain the level of safety otherwise provided by the design.

(2) The exemption is not inconsistent with the Atomic Energy Act or any other statute and therefore is authorized by law. As discussed above, the design change will not present an undue risk to the public health and safety. The design change does not relate to the common defense and security.

(3) Special circumstances are present as specified in 10 CFR 50.12(a)(2). Specifically, the remedial measure of water-tight doors provides a net increase in public safety relative to the design specified in the DCD, satisfying special circumstance (iv). Additionally, the changes qualify for special circumstance (ii) in that the changes are intended to accomplish the underlying purpose of the DCD, namely to ensure that the design is able to withstand natural phenomena. Further, special circumstance (vi) is present in that material circumstances not considered during the ABWR certification was granted in location and meteorological history analysis techniques. Given the need for power in Texas, it is in the public interest to allow construction of additional reactors at the STP site.

(4) The special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. Specifically, the design change of adding water-tight exterior doors represents an improvement in safety, and does not affect the configuration of the plant or the manner in which the plant is operated. Therefore, the reduction in standardization resulting from the change should not adversely affect safety.

As demonstrated above, this exemption complies with the requirements in Section VIII.A.4 of Appendix A to 10 CFR Part 52. Therefore, STPNOC requests that the NRC approve this exemption.

STD DEP 1.8-1, Tier 2* Codes, Standards, and Regulatory Guide Edition Changes

Description

Tier 2, Table 1.8-20 lists reference ABWR DCD compliance with NRC regulatory guides. Table 1.8-21 lists applicability of industry codes and standards. This departure identifies Tier 2* items on these two tables that are being updated to more current revisions/editions. Those Tier 2 items that are explicitly revised in the COLA or require change due to changes in the Tier 2* items are also included.

Regulatory Guide 1.75, “Physical Independence of Electric Systems,”~~,~~ Revision 3, dated 2/05; and Regulatory Guide 1.153, “Criteria for Power, Instrumentation, and Control Portions of Safety Systems,”~~,~~ ~~r~~Revision 1, dated 6/96 are adopted to ensure ~~b~~ more recent industry design and construction practices are used.

The 1992 edition of IEEE 384 “Criteria for Independence of Class 1E Equipment and Circuits” is adopted. IEEE 603 “Standard Criteria for Safety Systems for Nuclear Generating Stations” is updated to the 1991 version. These editions of the standards are currently endorsed by the NRC.

Mil-Specs for electromagnetic inference analysis and control are updated to more current versions as this field has advanced considerably since certification.

Current approved ASME code cases per Regulatory Guide 1.84, “Design and Fabrication Code Case,”~~,~~ ~~r~~Revision 33, dated 8/05 may be used in the future. With this update, Regulatory Guide 1.85, “Materials Code Case Acceptability, ASME Section III, Division 1" on ASME material code cases is obsolete and has been deleted as it is now incorporated into ~~r~~Revision 33 of R.G. 1.84.

The American Concrete Institute code ACI 349 is updated to the 1997 edition. The ASME Section III Division 2 is updated to the 2001 edition with 2003 Addenda. These combined recognize advances in earthquake engineering and allows efficient use of modularization during construction. Note that ASME Section III Division 1 for piping is not changed from the 1989 edition. This departure also updates Tier 2 to refer to Regulatory Guides 1.136, “Materials, Construction, and Testing of Concrete Containments,” ~~r~~Revision 3, dated 3/07, and Regulatory Guide1.142, “Safety-Related Concrete Structures for Nuclear Power Plants” to ~~r~~Revision 2, dated 11/01.

Evaluation Summary

As a Tier 2* departure, this departure requires prior NRC approval. These updates to more current revisions/editions will increase plant reliability and performance by capturing selected advancements in engineering theory and practice since issuance of the design certification. The revisions to the Regulatory Guides are the current ones in force. The revisions to the industrial codes and standards have been approved or endorsed by the NRC. These enhancements will provide the same level of plant protection and personal safety and are a net benefit to the public health and safety. Changes to Tier 2 items are incidental to the Tier 2* changes.